Exelon Generation

Exelon Generation Company, LLC
Braidwood Station
35100 South Route 53, Suite 84
Braceville, IL 60407-9619

www.exeloncorp.com

10 CFR 50.73

June 22, 2012
BW120061

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Braidwood Station, Unit 1
Facility Operating License No. NPF-72
NRC Docket No. STN 50-456

Subject: Licensee Event Report 2012-002-00 – Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzle Weld Indication Attributed to Primary Water Stress Corrosion Cracking

The enclosed Licensee Event Report (LER) is being submitted in accordance with 10 CFR 50.73, "Licensee Event Report System."

There are no regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact Mr. Chris VanDenburgh, Regulatory Assurance Manager, at (815) 417-2800.

Respectfully,

Daniel J. Enright

Daniel J. Enright
Site Vice President
Braidwood Station

Enclosure: LER 2012-002-00

cc: NRR Project Manager – Braidwood Station
Illinois Emergency Management Agency – Division of Nuclear Safety
US NRC Regional Administrator, Region III
US NRC Senior Resident Inspector (Braidwood Station)
Illinois Emergency Management Agency - Braidwood Representative

NRC FORM 366 (10-2010)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE
Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzle Weld Indication Attributed to Primary Water Stress Corrosion Cracking

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME: N/A	DOCKET NUMBER: N/A
04	23	2012	2012	- 002 -	00	06	22	2012	FACILITY NAME: N/A	DOCKET NUMBER: N/A

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
N/A – Defueled	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☒ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
000	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
Chris VanDenburgh, Regulatory Assurance Manager	(815) 417-2800

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO		N/A	N/A	N/A

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

During the Braidwood Unit 1 spring 2012 refueling outage, scheduled Volumetric Examination of the upper reactor pressure vessel (RPV) [AB] head penetrations was performed in accordance with the Electric Power Research Institute Performance Demonstration Program by the implementation of 10 CFR 50.55a and in accordance with American Society of Mechanical Engineers Section XI Code Case N-729-1 as amended by 10 CFR 50.55a(g)(6)(ii)(D).

On April 23, 2012, at approximately 0100, an indication was discovered in Penetration 69. The flaw was located on the outside diameter of the penetration tube and was axially oriented with a linear extent of 0.600 inches and a through wall depth of 0.216 inches, approximately 33.5% through wall.

The embedded flaw was subsequently repaired in accordance with NRC approved WCAP 15987, Revision 2-P-A, and WCAP 16401-P Revision 0. The apparent cause of the flaw is attributed to primary water stress corrosion cracking. Corrective actions included repairing the indication in Penetration 69 and revising the frequency of the Unit 1 Bare Metal Visual Exam and the Volumetric Exam to every refueling outage.

This event is reportable in accordance with 10 CFR 50.73(a)(2)(ii)(A), any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

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NARRATIVE

A. Plant Operating Conditions Before the Event:

Event Date: April 23, 2012

Unit: 1 MODE: N/A – Defueled

Unit 1 Reactor Coolant System [AB]: At ambient temperature and depressurized

No structures, systems or components were inoperable at the start of this event that contributed to the event.

Description of Event:

During the Braidwood Unit 1 spring 2012 refueling outage, scheduled Volumetric Examination of the upper reactor pressure vessel (RPV) [AB] head penetrations was performed in accordance with the Electric Power Research Institute Performance Demonstration Program by the implementation of 10 CFR 50.55a and in accordance with American Society of Mechanical Engineers (ASME) Section XI Code Case N-729-1 as amended by 10 CFR 50.55a(g)(6)(ii)(D).

On April 23, 2012, at approximately 0100, an indication was discovered in Penetration 69. The flaw was located on the outside diameter of the penetration tube and was axially oriented with a linear extent of 0.600 inches and a through wall depth of 0.216 inches, approximately 33.5% through wall.

A demonstrated volumetric leak path assessment was performed on all 78 of the control rod drive mechanism penetrations in accordance with 10 CFR 50.55a(g)(6)(ii)(D) and no indication of through wall leakage was observed. A Bare Metal Visual Inspection of the exterior surfaces of the reactor head and penetrations was performed in accordance with AME Section XI Code Case N-729-1, with no indication of through wall leakage observed.

The embedded flaw was subsequently repaired in accordance with NRC approved WCAP 15987, Revision 2-P-A, and WCAP 16401-P Revision 0.

Per the 2004 Edition of ASME Section XI Acceptance Criteria in Table IWB-3663-1 General Note (a), "Linear surface flaws of any size in the partial penetration nozzle to vessel (J-groove weld) are not acceptable." Therefore, this event is reportable in accordance with 10 CFR 50.73(a)(2)(ii)(A), any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

B. Cause of Event

The apparent cause of the flaw is attributed to primary water stress corrosion cracking (PWSCC).

The Braidwood Unit 1 reactor head penetration tubes are made of SB-167 (Inconel 600) material. The tubes were roll straightened after being annealed, which is suspected to have induced cold working into the material during the roll straightening process making these tubes more susceptible to PWSCC. The penetration tubes are attached to the reactor head by Inconel 82/182 weld filler material. Inconel 82/182/600 (Alloy 600) materials are recognized as being inherently susceptible to PWSCC, although the Braidwood Unit 1 reactor head was considered to be a low susceptibility head based on existing industry guidelines and regulations.

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NARRATIVE

This was the second Volumetric Examination performed on the Braidwood Unit 1 reactor head penetrations. The first Volumetric Examination was performed during the April 2006 refueling outage. No evidence of PWSCC was identified during the 2006 examination.

Prior to the Unit 1 refueling outage, the Volumetric Examination was every four outages and the frequency for the Bare Metal Visual Examination was every three outages. As a result of discovery of the indication in Penetration 69, both examinations are now required every refuel outage.

D. **Safety Consequences:**

This condition had no actual safety consequences impacting plant or public safety. The flaw was identified in a timely manner and repaired prior to through wall leakage occurring. The flaw was identified as part of a required periodic inspection. Potentially, if the flaw remained undetected, it could have over time propagated through the Alloy 600 weld material to form a leak path through the reactor coolant pressure boundary.

Based on the Unit 1 spring 2012 documented characteristics and dimensions of the flaw (axially oriented with a linear extent of 0.600 inches and a through wall depth approximately 33.5% through wall), there was no safety significant functional failure as a result of this flaw as no safety functions were lost. The primary coolant pressure boundary was maintained and capable of preventing the release of radioactive material. The rod drive system remained functional.

E. **Corrective Actions:**

- The indication in Penetration 69 was repaired prior to the return to service of the Unit 1 reactor head.
- The frequency of examination of Unit 1 for both the Bare Metal Visual Exam and the Volumetric Exam has been changed to every refueling outage.

F. **Previous Occurrences:**

No previous, similar Licensee Event Reports were identified at the Braidwood Station.

G. **Component Failure Data:**

Manufacturer	Nomenclature	Model	Mfg. Part Number
N/A	N/A	N/A	N/A